ACCESSION #: 9412200148
LICENSEE EVENT REPORT (LER)

FACILITY NAME: River Bend Station PAGE: 1 OF 18

DOCKET NUMBER: 05000458

TITLE: REACTOR SCRAM DUE TO SPURIOUS SIGNALS FROM UNDAMPED ROSEMOUNT MODEL 1153 TRANSMITTERS
EVENT DATE: 09/08/94 LER #: 94-023-01 REPORT DATE: 12/12/94

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 97

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(i), 50.73(a)(2)(iv), 10CFR21 SPL.RPT:T.S.3.5.1

LICENSEE CONTACT FOR THIS LER:
NAME: T.W. Gates, Supervisor-Nuclear Licensing TELEPHONE: (504) 381-4866

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: JC COMPONENT: LT MANUFACTURER: R370
B BN TRB D245
REPORTABLE NPRDS: Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On September 8, 1994 at 8:28 PM, with the reactor at 97 percent power, an automatic reactor scram occurred due to a false high reactor water level condition sensed on channels C and D of the reactor water level instrumentation. During this event, the RCIC turbine tripped due to binding of the turbine governor valve. The conditions leading to this failure have been determined to be reportable pursuant to 10CFR21. Since the HPCS system was manually operated during this event, this supplement also finalizes the Special Report required by Technical Specification 3.5.1 concerning emergency core cooling system (ECCS) injections.

The cause of this event is spurious signals from undamped Rosemount model 1153 transmitters in response to process noise. The model 1153 transmitters that were in service in the reactor water level

instrumentation application have been replaced with Rosemount model 1152s. Extensive monitoring was conducted as a conservative measure during the startup from the forced outage and continuing into power operation for a limited period of time.

The investigation of transmitter performance revealed that the model 1153 susceptibility to process noise would not have prevented the transmitters from functioning properly in an actual event. Equipment and radiological issues, including reactor vessel cooldown and the Technical Specification surveillance time limit non-compliances for radiological and chemistry sampling were evaluated and determined not to be safety significant. Therefore, this event did not compromise the health and safety of the public.

END OF ABSTRACT

1.0 REPORTED CONDITION

On September 8, 1994 at 8:28 PM, with the reactor at 97 percent power, an automatic reactor scram occurred due to a false high reactor water level condition sensed on channels C and D of the narrow range reactor water level instrumentation (*JC-LT*). During the course of the event, surveillance time limits requiring sampling of noble gases, tritium, and reactor coolant conductivity were not met. Therefore, this event is reported pursuant to 10CFR50.73(a)(2)(iv), to document the reactor scram, and 10CFR50.73(a)(2)(i)(B) to document the non-compliances with the Technical Specifications.

2.0 INVESTIGATION

2.1 Initial Conditions

The plant was at 97 percent power with power ascension in progress to 100 percent power at a rate of 1 percent per hour. During the previous shift, power had been reduced to 76 percent in response to loss of a non-safety-related chiller. No surveillance test procedures were being performed and no maintenance was in progress in the containment.

2.2 Event Description

On September 8, 1994, at 8:28 PM, an automatic reactor scram occurred due to a false high reactor water level condition, sensed by the C and D channels of the narrow range reactor water level instrumentation. The control room operators had no indication of the origin of the scram at the time it occurred. There was no control room indication of a reactor water level increase or a feedwater level excursion. Operators initiated recovery procedures.

By design, the reactor scram did not result in an automatic trip of the main turbine (*TA*) or electric generator (*TB*) or the reactor feed pumps (*SJ-P*). During the process of completing AOP-0002, "Turbine/Generator Trip," the unit operator (UO) recognized that the turbine had not tripped. Recognizing that the normal trip for this condition would be the generator trip on reverse power, the operator attempted to determine if a reverse power condition actually existed. The digital generator load indicator was alternately indicating 5 and 6 MW. The analog generator load indicator had decreased to 0 MW, but the generator output breakers (*TB-BKR*) had not opened on reverse power as expected by the operator. The UO immediately reported to the Control Room Supervisor (CRS) that the turbine was still on-line.

Since reactor pressure was continuing to drop, the crew felt that some action was required to take the turbine off-line regardless of whether or not a reverse power condition existed. After evaluating the condition, the CRS directed the crew to manually trip the turbine, intending to intentionally arm the generator anti-motoring trip function, so that the generator output breakers would automatically open.

Following the turbine trip, the main generator failed to trip on reverse power and was manually tripped at 8:40 PM, approximately twelve minutes after the reactor scram. The manual trip of the generator resulted in a slow bus transfer of non-safety related station services, as designed.

The slow bus transfer resulted in the de-energization of non-safety related loads as the bus supply source was shifted from the normal station service transformers (*XFMR*) to the preferred station service transformers (*XFMR*) (i.e., off-site power). The de-energization of the non-safety related buses resulted in the loss of power to all condensate pumps (*SD-P*), all feedwater pumps (*SJ-P*), reactor recirculation pumps (*AD-P*), and both Reactor Protection System (RPS) (*JC-BU*) buses. Loss of normal power to the RPS buses caused a balance of plant isolation and main steam isolation valve (ISV) closure. This loss of electrical power also caused a failure of the Safety Parameter Display System (SPDS) (*IU*) and the Emergency Response Information System (ERIS) (*IQ*) computers.

The Reactor Core Isolation Cooling (RCIC) (*BN*) system was manually started to provide make-up to the reactor pressure vessel, but tripped on a mechanical overspeed condition. The High Pressure Core Spray (HPCS) pump (*BG-P*) was then manually started and used to raise Reactor Pressure Vessel (RPV) level and maintain adequate core cooling. Main steam safety-relief valves (*SB-RV*) were cycled by the operators, as required by procedures, to control RPV pressure. During the event, an automatic transfer of the HPCS suction source, from the condensate storage tank (CST) (*TK*) to the suppression pool, occurred on high suppression pool water level. After due consideration, the HPCS system was manually transferred back to the CST, as directed by EOP-0001 "RPV Control."

Emergency procedures were utilized to assure control of RPV and containment parameters. On three occasions, SRVs automatically actuated at the relief setpoint. At 10:09 PM, the Shift Superintendent declared a Notification Of Unusual Event (NOUE) at his discretion to mobilize assistance to maintain the plant in a stable condition. There were no unmonitored radiological releases and all effluents remained within established limits.

At 11:21 PM, reactor feedwater was restored to service. Restoration of other plant systems was proceeding in accordance with plaint procedures. At 12:30 AM on September 9, all Emergency Operating Procedures were exited and the NOUE was terminated.

2.3 SEQUENCE OF EVENTS

20:28 Automatic reactor scram (Initiating signal: RPV Water Level 8 signals to RPS channels C and D.).

Recirculation pumps transferred to slow speed automatically.

20:38 Manual trip of main turbine.

20:40 Manual trip of Main Generator output breakers.

Normal (13.8 kV) station service buses NPS-SWG1A and NPS-SWG1B "slow transfer" from the normal station service transformers to the preferred station service transformers. Non-safety related plant equipment was deenergized as follows:

- Condensate and feedwater pumps (loss of normal high pressure makeup to the reactor vessel).

- RPS A and B (results in a full MSIV and BOP isolation). (Normal power supply to safety related RPS busses is via non-safety related motor generator sets. RPS fails safe on loss of power.)

- Reactor recirculation pumps.

- Circulating water pumps A & C.

- Instrument Air Compressor B

- One Normal Service Water pump.

- Emergency Response Information System (ERIS) computer.

- Safety Parameter Display System (SPDS) computer.

20:41 RPS A&B manually transferred to alternate supply.

20:44 Operators attempted to provide coolant makeup water to the reactor via the Reactor Core Isolation Cooling (RCIC) System. The RCIC turbine trips on overspeed and cannot be reset from the Main Control Room.

Safety Relief Valves used to manually control reactor pressure.

20:49 Restored Drywell Cooling.

20:57 High Pressure Core Spray (HPCS) pump started manually to provide coolant makeup water to the reactor. Level at 0" (wide range) and lowering (Note: Normal operating water level is +35 inches, auto-initiation setpoint is -43 inches, and the top of the active fuel is -162 inches).

21:18 Opened B21*MOVF019 (*SB-20*), Main Steam Drain Outboard Isolation Valve, establishing a vent path from the reactor vessel to the main condenser to assist in reactor pressure control.

21:20 Restored Turbine Building Chillers (*NM-CHU*) to service.

21:27 Started Residual Heat Removal System in Suppression Pool Cooling Mode.

21:38 Valve 1CNS-MOV112 (*SD-20*) could not be opened during condensate fill and venting

21:56 Reset Reactor Scram.

22:03 Re-inserted one-half scram on Division I to comply with Technical Specification 3.3.1, "Reactor Protection System Instrumentation.

22:09 Notification of Unusual Event declared.

22:20 Restarted Condensate Pump CNM-P1A (*SD-P*).

23:21 Started Main Feed Pump A (*SJ-P*).

23:51 Re-opened Main Steam Isolation Valves (*SB-ISV*) after chillers reduced area temperatures below the isolation setpoint.

00:17 Secured HPCS. Reactor water level maintained with main feed pump.

00:30 Exited Emergency Operating Procedures and terminated Notice of Unusual Event.

2.4 Turbine Response

As designed the reactor scram did not result in an automatic trip of the main turbine. Instead, operators manually tripped the turbine at 2238, ten minutes after the scram. Operators manually tripped the generator breakers at 2040. The manual trip of the generator resulted in a slow bus transfer of nonsafety-related station services, as designed.

The feedwater control system reactor vessel level transmitters are used to sense reactor water level and trip the main turbine and feedwater pumps on high water level. The nuclear boiler instrumentation reactor vessel level transmitters sense reactor water level and trip the reactor on high water level. In this case, since two level transmitters in the nuclear boiler instrumentation system sensed the high reactor water level, an automatic scram resulted. However, since only one level transmitter in the feedwater control system sensed a high reactor water level, the main turbine and feedwater pumps did not automatically trip. Process computer data indicate that the scram was caused by level 8 signals from narrow range reactor water level instrumentation channels C and D. ERIS data indicates that narrow range feedwater level transmitter 4C reached the level 8 setpoint and that 4A and 4B did not. The two-out-of-three logic required to produce a turbine trip was not satisfied since only one of three channels reached the level 8 setpoint. Therefore, with regard to the reactor vessel high water level signals, the main turbine trip logic functioned as designed.

2.5 Generator Response

By design, the reactor scram did not result in an automatic trip of the main turbine or generator. Operators inserted a manual trip of the turbine approximately 10 minutes after the reactor trip. The manual turbine trip resulted in turbine stop valve closure. Following the turbine trip, the main generator did not trip on reverse power. Normally, the generator output breakers are expected to open upon reverse power to the generator following a reactor scram. The generator output breakers were manually opened at 2040, approximately twelve minutes after the reactor scram, since the reverse power trip function had not initiated. The manual trip of the generator resulted in a slow bus transfer of non-safety related station services, as designed.

The investigation revealed that the failure of the reverse power trip to initiate as expected was due to common mode calibration inaccuracies in the reverse power relays, 32G and 32G1, combined with a very low power factor (i.e., high reactive load). The generator was operating under a large reactive load at a very low power factor which resulted in an extreme phase angle at the relay. The relays were found to have been misadjusted by 2 degrees for relay 32G1 and 4 degrees for relay 32G. This combined with inherent relay inaccuracy, resulted in the failure of the relays to actuate because the generator was operating within the error band of the relay trip point. This is the root cause of the failure of the generator output breakers to open on reverse power.

2.6 Transfer to Offsite Power

During a main turbine trip, the main generator should trip after reverse power occurs. Two automatic transfer schemes ("fast" and "slow") are provided to transfer station electrical loads from the main generator to off-site power. In accordance with the system design, a slow, instead of a fast, transfer occurred during this event. A slow bus transfer provides a protective function for station equipment and differs from a fast transfer in that it results in the tripping of all bus loads. Manual restoration of those loads is required following a slow transfer.

The slow transfer of 1NPS-SWG1A and 1B was not anticipated by Operations personnel, but the evaluation revealed that it occurred correctly. Since the generator output breakers were manually tripped prior to the reverse power trip occurring, relay logic blocked the fast transfer

from occurring. Thus, the prerequisites for the fast transfer were not met. With regard to the function of the fast/slow transfer circuits, no corrective action is required. However, the indications available to the operators could be improved to allow evaluation of the reverse power condition and support operators' decision when to trip the generator output breakers.

2.7 RCIC Turbine Trip

On September 8, 1994, subsequent to the manual opening of the generator output breakers after the scram, the slow transfer to the preferred offsite power resulted in a loss of normal feedwater.

Upon the loss of feedwater, the operators initiated actions to manually start the RCIC turbine in anticipation that it may be needed to help control reactor vessel coolant level and reactor pressure. The RCIC turbine tripped when steam was admitted to the turbine. The operator could not reset the RCIC turbine from the control room and the indications that he had were consistent with a mechanical overspeed trip which by design must be reset locally. Subsequent field investigation verified that the mechanical overspeed trip device was actuated and had caused the RCIC turbine to trip. The cause of the RCIC pump turbine overspeed was found to be binding of the turbine governor valve due to accelerated corrosion of the valve stem. The root cause of the accelerated corrosion is the combined effect of problems with the surface treatment of the governor valve stem, improper washer material in the valve gland area and characteristics of the carbon spacers in the gland area (i.e., porosity and the presence of sulfur). The investigation revealed that the surface treatment of the stem was non-uniform, with variations in thickness and defects present. The sulfur in the carbon spacers can leach out in a moist environment and create an electrolytic solution to support galvanic corrosion. The improper washer material can also promote galvanic corrosion. EOI has determined that this condition is reportable pursuant to 10CFR21. The stem, washers and spacers were manufactured by Terry Steam Turbine Company. Dresser-Rand Steam Turbines is the current vendor. The stem, spacers, and washers were new equipment installed during refueling outage 5.

The washers supplied in 1984 were installed during refueling outage 5. One of these washers was selected for analysis which revealed that it was made out of 300 series stainless steel instead of 410 stainless. Another group of washers was supplied in 1985. Of the 21 washers in the 1985 order, 20 of them were 300 series stainless steel, and one was 400 series stainless steel. The part number of the washers supplied in 1984 and 1985 was the same, P/N#54846.

2.8 MOV Issues

The post-scram investigation revealed that SWP*MOV40A (*BS-20*) failed during midstroke due to a short in one of its control cables. The safety function of 1SWP*MOV40A is to open during a standby service water initiation. Valve 1SW*PMOV40A was approximately 30% open when it failed during mid-stroke. A generic design vulnerability applicable only to Limitorque SMB-00 actuators was identified and measures have been implemented to prevent recurrence.

In addition, several non-safety power operated valves (MOVs and SOVs) also failed to respond as expected. These valves were in balance-of-plant (BOP) systems and had no impact on the ability to safely shut down the reactor and maintain it in a safe shutdown condition.

The root cause for the problems associated with the non-safety related valves is the lack of a preventive maintenance program.

2.9 Event Response Information System and Safety Parameter Display System

During the plant transient, the normal power supply to the

Safety Parameter Display System (SPDS), transient analysis computers which is part of the Emergency Response Information System (ERIS) and Digital Radiation Monitoring System (DRMS) was lost. Upon discovering that the computer systems were inoperable, the system engineer attempted to archive any available data, then restarted the computer systems and restored them to their normal display and data collection functions. The cause of the failure was that the power inverter (*INVT*), 1BYS-INV06, which supplies power to these systems, was unavailable. The inverter was in bypass for maintenance.

2.10 Reactor Vessel Stratification, Cooldown, Pressure/ Temperature Limits

The investigation included evaluation of reactor vessel stratification, cooldown, and the effect on pressure and temperature limits. The cooldown rate exceeded the Technical Specification limit of 100 degrees F per hour. The evaluations to address these issues revealed that in each case, the thermal transient effects were bounded by previous analyses, including the thermal transient effects due to the cooldown rate. Usage factors for the HPCS nozzle, piping, and recirculation system piping and components were determined to be within the design values. The total accumulated actuation cycles for the HPCS nozzle was calculated to be 15. The circumstances that led to the initiation of the HPCS system are described in Section 2.2, Event Description. This report provides the information required for the Special Report pursuant to T. S.3.5.1.

2.11 Noble Gas and Tritium Samples

After the reactor scram, Chemistry did not obtain samples of main plant noble gas and tritium within one hour even though the dose equivalent I-131 concentration exceeded three times normal. The tritium and noble gas samples were taken approximately one hour late.

Following the event, an investigation of the TS requirements was conducted. This investigation found that the TS wording changed prior to issue of the initial low power operating license to add the one hour time limit for sampling tritium and noble gases following thermal transients. The change created a time requirement that is inconsistent with the other licensing basis documents reviewed and the TS from the other operating boiling water reactor (BWR) 6 plants in the United States. The one hour limit following reactor thermal transients cannot be fulfilled following a reactor scram due to time requirements for sampling and analysis. While the surveillance was not performed within one hour, the requirements of the action statement of T.S.3.11.2.1 were not violated. The dose rate due to radioactive effluents was always within the TS limits.

The missed sample was a recurrence of a previous event, documented in LER 87-013 and Condition Report (CR) 87-962, in which the same TS samples were missed following a reactor scram. In that event, the root cause was failure of control room personnel to notify chemistry personnel that the plant had scrammed. The corrective actions for that event included adjusting the volume on the plant paging system in the chemistry lab and investigating a possible change to the TS. The response from that investigation stated that there was inadequate justification to request a change. The corrective actions for LER 87-013 were not sufficient to prevent recurrence and are considered part of the cause of the missed chemistry sample.

Contributing factors included absence of the sample pump at 1RMS*RE125, and delays entering the Auxiliary Building due to operation of the SGTS.

2.12 Conductivity Sample

Following the reactor scram, chemistry failed to obtain the reactor coolant conductivity analysis once per every four hours after a loss of continuous conductivity recording. Prior to the reactor scram only the Reactor Water Cleanup System (*CE*) (WCS) influent conductivity monitor was operable in accordance with TS 3/4.4.4. The recorder in the control room for the reactor recirculation conductivity monitor had been determined to be inoperable earlier that day by the on-shift chemistry technician. While obtaining the dose equivalent I-131 samples at 0206 of that same night the on-shift chemistry technician observed flow from the WCS sample line, although at a reduced rate. Communications with control room personnel at 0230 informed him that the WCS pumps had tripped following the scram; however, he was unaware that containment isolation valves for this system had closed and that the reactor recirculation conductivity recorder was not operable.

The root cause of the missed conductivity sample was determined to be the lack of timely communications between control room and chemistry personnel regarding status of the reactor water cleanup system. Chemistry personnel were also unaware that the reactor recirculation conductivity recorder was inoperable.

2.13 Radiological Impact

Two radiological transients occurred subsequent to the scram. A transient in the turbine building ventilation system resulted in a build-up of noble gases in the turbine building. After the ventilation system was restored to service, noble gas levels rapidly decreased to normal. In addition, a radiological transient in the containment building occurred subsequent to safety relief valve actuation which resulted in an increase in containment building activity. An evaluation and off-site dose calculation was performed prior to initiating a reactor building purge. As a result, radiological conditions in containment stabilized and returned to normal.

The contribution of these transients to the off-site dose was below TS and 10CFR off-site radiological limits. A review of the events determined that the radiological procedures utilized during the event were adequate for transient events. The review also concluded that communication and staffing (including augmented staffing) were adequate to perform the required RP activities. No corrective actions are required.

3.0 Root Cause Evaluation

All available data associated with reactor operation that could potentially affect reactor water level instrumentation was reviewed and all potential failure modes were identified using event and causal factors charts, Kepner-Tregoe (K-T) analysis, and failure mode analysis.

Two major paths were considered in the investigation of the level 8 signal. One of these paths considered an actual change in reactor vessel level. The other path considered was an indicated level transient. The analysis of the events in the indicated level transient path led to the conclusion that the probable cause of the event was process noise resulting in a large amplitude trip signal on the RPS C and D level transmitters and feedwater level transmitter C. The investigation included in-vessel-visual-inspections (IVVI). The information gained from these inspections was evaluated and resulted in ruling out many theorized causes.

The cause of this event is spurious signals from undamped Rosemount model 1153 transmitters in response to process noise. All three of these transmitters are Rosemount model 1153 transmitters. Rosemount model 1152 transmitters were used for RPS channels A and B and these channels did not initiate a level 8 signal. The investigation revealed that all three of the model 1153 transmitters had been installed as replacements for Rosemount model 1152s.

The three affected 1153s had minimum damping; two were set at minimum damping and one had no damping card installed. The investigation of the damping issue revealed that the time response testing requirements for the transmitters results in minimal damping.

The investigation also revealed deficiencies in the maintenance of these transmitters. While these issues did not contribute to the root cause, they are being addressed. A damping card was not installed on RPS level channel C and feedwater level transmitter C was undamped. However, if the damping card had been installed on RPS channel C, it would probably have been set to minimum damping, and the scram would still have occurred. The minimization of damping was permissible given the design guidance available to maintenance personnel; however, improvements in the areas of generic modification guidance and maintenance planning win be evaluated.

Based on testing that was performed, engineering personnel concluded that the transmitters would have functioned properly during an actual level transient. The investigation also revealed that no electrical or significant hydraulic transient existed.

4.0 Corrective Action

As a result of the September 8 event, Entergy Operations promptly formed a "Significant Event Response Team" (SERT) to investigate the event and develop appropriate corrective actions. The SERT team was authorized by the plant manager and its membership included a high level of management from multiple departments. The team's function was to investigate root cause and provide corrective actions for all deficiencies identified during the September 8 event. Management oversight was provided by members of the executive staff led by John McGaha, Vice President-Operations.

The event response organization was supplemented by offsite Entergy Operations personnel and nuclear industry expertise, including General Electric and root cause analysis experts from Failure Prevention International (FPI). An assist team from the Institute of Nuclear Power Operations (INPO) was also onsite to investigate the event.

Review of selected condition reports associated with this event was conducted by the Corrective Action Review Board (CARB). This board is comprised of the direct reports to the Vice President - Operations, the General Manager - Plant Operations and his direct reports, Manager, Nuclear Safety and Assessment, and the QA Manager. This review is conducted to assure proper root cause determination and development of effective corrective actions for events determined to be significant by the criteria of River Bend Nuclear Procedure RBNP-030, "Initiation and Processing of Condition Reports. "

The sections below document the current status of the primary corrective actions for the issues identified in this event.

4.1 Rosemount Model 1153 Transmitters and Backfill System

o The Rosemount 1153 transmitters that were in service in the reactor water level instrumentation and feedwater level applications have been replaced with Rosemount model 1152s which do not have the same sensitivity to process noise.

o A verification of all aspects of the configuration of all safety related Rosemount transmitters was performed prior to startup. Plant walkdowns were used to baseline the configuration and verify the transmitters based on model number, required damping, and mounting.

o Time response testing methodology will be reviewed with a focus on industry practices.

o Generic modifications for changeouts of equipment and the maintenance planning process will be evaluated.

o To address a potential vulnerability identified by the investigation, the backfill system has been modified to relocate the orifices downstream of the check valves.

o EOI developed a monitoring program to track important process parameters during the startup from the forced outage and following this for a limited time during power operation. The objective of this program was to identify operational anomalies to minimize the risk of recurrence, as a conservative measure. The monitoring program was completed with no unusual events or anomalies detected.

4.2 Operations

With respect to operator performance, several lines of investigation are being pursued as a result of this event. The goal of this investigation is to identify areas where enhancements will result in improved operator performance. Specific areas of interest include:

o Event Reconstruction. In the interest of obtaining a complete, clear understanding of a significant plant event, Operators should be debriefed as soon as possible. Although individual debriefings were conducted by operations management, a full crew debriefing was not conducted in a timely manner. The delay in conducting a full crew debriefing will be evaluated and appropriate guidance developed regarding the timeliness of these interviews.

o Procedures. The AOP for turbine and generator trip contains requirements related to verification of generator trip. This procedure, AOP-0002, has been revised to improve the procedural guidance for positive verification of reverse power conditions. Procedure Enhancements identified during review included revision of AOP-0001, "Reactor Scram," to improve the turbine trip verification, and SOP-0080, "Turbine Generator Operation," to provide a caution on turbine/generator motoring.

o Training. The crew's understanding of the issue of the fast/slow transfer of station loads was not clear and the simulator modeling and associated training was incorrect. Simulator modifications have been implemented to correct deficiencies. Training has been provided during the last licensed operator requalification module concerning the procedure changes to AOP -0001 and AOP-0002. In addition, a simulator scenario has been developed which requires operator action to manually open the generator output breakers following failure of the generator reverse power/anti-motoring trips.

4.3 Generator Response

Both reverse power relays were recalibrated to maintain the phase angle of each at its setpoint with the tightest tolerance attainable. Improvements in the applicable maintenance procedure, MCP-1005, are being considered.

4.4 Transfer to Offsite Power

To improve the indications available to the operators for evaluation of the reverse power condition and determining when to trip the generator output breakers, the SPDS system graphic display in the control room has been upgraded to indicate negative megawatts. This display will allow operators to monitor reverse power conditions.

4.5 RCIC Turbine Trip

The governor valve stem has been replaced with a new stem having an aluminized coating for increased corrosion resistance. Washers of the proper material have been installed, and periodic monitoring of the stem resistance is being performed, pending further evaluation of monitoring data.

4.6 Motor Qperated Valves

Corrective actions being implemented for SWP*MOV40A are:

o The damaged wire and lug were replaced and repositioned to avoid rubbing.

o Nine (9) additional SMB-00 actuators were identified and have been inspected for similar lug configurations on contacts LS-1 and LS-9. No additional problems were identified.

o Maintenance procedures will be revised to include guidance on proper positioning of wires landed on contacts LS-1 and LS-9.

River Bend Station is implementing a preventive maintenance program action plan with a focus on reliability centered maintenance (RCM), and prioritization by Maintenance Rule system and component importance. The predictive and preventive maintenance tasks for non-safety related valves will be addressed in the context of this program.

4.7 ERIS and SPDS

The services building power inverter, 1BYS-INV06 has been restored to service. Replacement of the ERIS system is being evaluated. This evaluation will also address concerns with the ease of retrieval of historical data from past events.

4.8 Noble Gas and Tritium Sampling

To prevent recurrence, Technical Specifications 3/4.11.2.1.2, Table 4.11.2.1.2-1 will be revised to remove the one hour sampling and analysis requirement for noble gases, and the tritium sampling requirements. License Amendment Request (LAR) 94-11, "Gaseous Effluents, " was submitted to the NRC on October 4, 1994 (RBG-40919). Other corrective actions include changes to operations announcement practices, revision of SOP-0043 to provide safe access to the auxiliary building when the standby gas treatment system is in operation, and ensuring the proper equipment is dedicated and staged for ready access near 1RMS*RE125. These actions have been implemented.

4.9 Conductivity Sample

Chemistry Procedure, CSP-0101, has been revised to incorporate a shutdown enclosure in the procedure. Corrective actions have also been implemented to address timeliness of required chemistry actions and assure that chemistry personnel coming on-shift will be cognizant of current equipment status.

5.0 Safety Assessment

Based on testing that was performed, engineering personnel concluded that the transmitters would have functioned properly during an actual level transient. The investigation also revealed that no electrical or significant hydraulic transient existed.

The evaluation of other equipment related issues revealed the following:

o The reactor scram did not result in an automatic trip of the main turbine or electric generator, by design. The "two out of three" logic required to produce an automatic turbine trip was not satisfied since only one of three feedwater level transmitter channels provided a level 8 signal.

o The slow transfer was also determined to have occurred as designed. The conditions required for a fast transfer to occur were not satisfied.

o The HPCS system was available throughout this event and was operated manually to provide makeup to the reactor vessel following the trip of the RCIC turbine.

o The reactor vessel cooldown rate has been evaluated and the thermal transient effects were bounded by previous analyses. Other thermal effects, such as thermal stratification, were also shown to be bounded by previous analyses.

o The contribution to offsite dose as a result of this event was analyzed and determined to be below Technical Specification limits and other regulatory limits.

Operator actions were correctly prioritized throughout the event. While they did encounter unexpected responses from some plant equipment, the operators effectively utilized the available resources to diagnose and respond to reactor and plant system indications. They focused on reactor safety and took actions to manually control reactor water level and pressure. Based on the above considerations, EOI concludes that this event did not compromise the health and safety of the public.

Note: Energy Industry Identification System (EIIS) Codes are identified in the text as (*XX*).

ATTACHMENT TO 9412200148 PAGE 1 OF 2

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Nuclear Safety

December 12, 1994

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Stop P1-37
Washington, DC 20555

Subject: River Bend Station - Unit 1
Docket No. 50-458
License No. NPF-47
Licensee Event Report 50-458/94-023-01
File No.: G9.5, G9.25.1.3

RBG-41099
RBF1-94-0129

Gentlemen:

In accordance with 10CFR50.73, enclosed is the subject report.

Sincerely,

JJF/jr
enclosure

ATTACHMENT TO 9412200148 PAGE 2 OF 2

Licensee Event Report 50-458/94-023-01
December 12, 1994
RBG-41099
RBF1-94-0129
Page 2 of 2

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*** END OF DOCUMENT ***